Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.73

May 18, 2004
3F0504-02

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: LICENSEE EVENT REPORT 50-302/04-001-00

Dear Sir:

Please find enclosed Licensee Event Report (LER) 50-302/04-001-00. The LER discusses actuation of the Reactor Protection System and Emergency Feedwater System caused by a failed circuit board within the Main Feedwater Integrated Control System on March 24, 2004. This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A).

No new regulatory commitments are made in this letter.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Jon A. Franke
Plant General Manager
Crystal River Nuclear Plant

JAF/dwh

Enclosure

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

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NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Reactor Trip Caused By Failed Circuit Board In The Main Feedwater Integrated Control System

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	24	2004	04	- 001	- 00	05	18	2004	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	1
10. POWER LEVEL	100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Dennis W. Herrin, Lead Engineer	(352) 563-4633

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	JA	90	B045	Y					

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

16. **ABSTRACT** (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

At 03:31, on March 24, 2004, Progress Energy Florida, Inc., (PEF) Crystal River Unit 3 (CR-3) was in MODE 1 (POWER OPERATION) at 100 percent RATED THERMAL POWER. While in steady state operation, a large decrease in the reactor demand and Main Feedwater flow demand signals occurred within the Integrated Control System (ICS). The Main Feedwater pumps ran back until discharge pressure was less than steam generator pressure, at which time the Main Feedwater flow reduced to zero. Upon sensing less than 17 percent Main Feedwater flow, the Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) initiated a turbine trip and actuated the Emergency Feedwater System (EFW). The Reactor Protection System (RPS) sensed the turbine trip above 45 percent reactor power and initiated an anticipatory reactor trip. The cause for this event was age-related failure of the zener diodes in the +15 volt regulator circuit for a Bailey 820 Control Module in the Main Feedwater Integrated Control System. The circuit board was replaced. RPS and EFW valid actuations are reportable under 10CFR50.73(a)(2)(iv)(A). This condition does not represent a reduction in the public health and safety. No previous similar occurrences have been reported to the NRC by CR-3.

17. TEXT *(If more space is required, use additional copies of NRC Form 366A)*

EVENT DESCRIPTION

At 03:31, on March 24, 2004, Progress Energy Florida, Inc., (PEF) Crystal River Unit 3 (CR-3) was in MODE 1 (POWER OPERATION) at 100 percent RATED THERMAL POWER. While in steady state operation, a large decrease in the reactor demand and Main Feedwater [SJ] flow demand signals occurred within the Integrated Control System (ICS) [JA]. As expected, Main Feedwater flow decreased faster than reactor power. The Main Feedwater pumps [SJ, P] ran back until discharge pressure was less than steam generator [AB, SG] pressure, at which time the Main Feedwater flow reduced to zero percent. Upon sensing less than 17 percent Main Feedwater flow, the Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) [JE] initiated a turbine trip and actuated the Emergency Feedwater System (EFW) [BA]. The Reactor Protection System (RPS) [JC] sensed the turbine trip above 45 percent reactor power and initiated an anticipatory reactor trip.

No structures, systems or components were inoperable at the start of the event that contributed to the event. Plant safety systems responded as expected during the reactor trip with the following exceptions:

> Auxiliary Steam Valve ASV-26 [SA, FCV] did not control auxiliary steam pressure in automatic and was placed in manual. The low Auxiliary Steam demand during this transient is outside the normal control range for ASV-26.
>
> Heater Drain Valve HDV-83 [SJ, V] did not adequately control the Deaerator Feed Tank (DFT) [SJ, DEA] level. This may have been caused by mis-calibration of level switch FW-6-LS [SJ, LS]. Condensate pumps CDP-1A and CDP-1B [SD, P] tripped on high DFT level. CDP-1B was restarted.

RPS and EFW valid actuations are reportable to the NRC. At 04:39, on March 24, 2004, a non-emergency four-hour notification and a non-emergency eight-hour notification were made to the NRC Operations Center (Event Number 40608) in accordance with 10CFR50.72(b)(2)(iv)(B) and 10CFR50.72(b)(3)(iv)(A), respectively . This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A).

SAFETY CONSEQUENCES

Based on the loss of Main Feedwater, valid actuation of AMSAC, EFW and RPS occurred as expected to shut down the reactor and maintain adequate steam generator levels. When the Main Feedwater flow decreased, AMSAC initiated a turbine trip and actuated EFW upon sensing less than 17 percent Main Feedwater flow. The RPS sensed the turbine trip above 45 percent reactor power and initiated a reactor trip. Reactor operators properly executed the Emergency Operating Procedures for plant shutdown.

Based on the above discussion, PEF concludes that actuation of the RPS and EFW did not represent a reduction in the public health and safety. This event does not meet the Nuclear Energy Institute definition of a Safety System Functional Failure (NEI 99-02, Revision 2).

CAUSE

The cause for this event was age-related equipment failure. Module 3-8-4 [JA, IMOD] in the Main Feedwater ICS is a Bailey 820 Control Module. The +15 volt regulator circuit on the Module 3-8-4 circuit board contained four failed devices. The five elements of this circuit are three 5 volt zener diodes connected in series to provide a regulated 15 volt supply, a capacitor to reduce ripple, and a current limiting resistor. The first failure was one of the 5 volt zener diodes. This failed to a shorted state. This failure forced each of the other devices to increase its load current, dissipating more heat. The failure of this first device may have occurred months before the next step occurred. The increased heat dissipation led to subsequent failure of the remaining circuit components. The cause for this event was failure of the first 5 volt zener diode due to end of life.

The Progress Generation Nuclear Electronic Service Center recommends replacing zener diodes, carbon resistors and operating-amplifiers when service has reached 25 to 30 years. This is in line with the guidance from Electric Power Research Institute (EPRI) Technical Report 1007916, "Printed Circuit Board Maintenance, Repair and Testing Guide."

CORRECTIVE ACTIONS

1. CR-3 Administrative Instruction AI-704, "Reactor Trip Review and Analysis," was performed.

2. ICS Module 3-8-4 was replaced with a bench-tested, calibrated module.

3. Other suspected ICS modules were pulled, inspected for visible damage and calibration checked on the test rack. No anomalies were identified.

4. Other actions associated with this event are being addressed in the CR-3 Corrective Action Program in Nuclear Condition Report 122486.

PREVIOUS SIMILAR EVENTS

No previous similar events involving failure of the Bailey 820 Multiplier Control Module 15 volt regulator circuit have been reported to the NRC by CR-3.

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ATTACHMENTS

Attachment 1 - Abbreviations, Definitions, and Acronyms

Attachment 2 - List of Commitments

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ATTACHMENT 1

ABBREVIATIONS, DEFINITIONS AND ACRONYMS

AI	Administrative Instruction
ATWS	Anticipated Transient Without Scram
AMSAC	ATWS Mitigation System Actuation Circuitry
ASV	Auxiliary Steam Valve
CDP	Condensate Pump
CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
DFT	Deaerator Feed Tank
EFW	Emergency Feedwater
EPRI	Electric Power Research Institute
HDV	Heater Drain Valve
ICS	Integrated Control System
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PEF	Progress Energy Florida, Inc.
RPS	Reactor Protection System

NOTES:

Improved Technical Specifications defined terms appear capitalized in LER text {e.g., MODE 1}

Defined terms/acronyms/abbreviations appear in parenthesis when first used {e.g., Reactor Building (RB)}.

EIIS codes appear in square brackets {e.g., reactor building penetration [NH, PEN]}.

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17. TEXT *(If more space is required, use additional copies of NRC Form 366A)*

ATTACHMENT 2

LIST OF COMMITMENTS

The following table identifies those actions committed to by PEF in this document. Any other actions discussed in the submittal represent intended or planned actions by PEF. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing & Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

RESPONSE SECTION	COMMITMENT	DUE DATE
	No regulatory commitments are being made in this submittal.	